2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 55.2 Kg/cm^2g (785 psig) or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 55.2 Kg/cm^2g (785 psig) and core flow ≥ 10% rated core flow:

(space)

MCPR shall be ≥ 1.07.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be maintained ≤ 93.1 Kg/cm^2g (1325 psig).

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the [General Manager—Nuclear Plant and Vice President—Nuclear Operations] and the [offsite reviewers specified in Specification 5.5.2, "[Offsite] Review and Audit"].

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (General Electric Corporation (GE) Fuel) (continued)

independent of bundle power and has a value of .246 Kg/cm^2 (3.5 psi). Thus, the bundle flow with a .316 Kg/cm^2 (4.5 psi) driving head will be > 12.7 m^3/hr (28 x 10^3 lb/hr). Full scale ATLAS test data taken at pressures from 1 Kg/cm^2a (14.7 psia) to 56.2 Kg/cm^2a (800 psia) indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 55.2 Kg/cm^2g (785 psig) is conservative.

2.1.1.2 MCPR (GE Fuel)

The fuel cladding integrity SL is set, such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, [1971 Edition] [later Edition], including Addenda through the [winter of 1972] [later Edition] (Ref. 5), which permits a maximum pressure transient of 110%, 96.7 Kg/cm^{2}g (1375 psig), of design pressure 87.9 Kg/cm^{2}g (1250 psig). The SL of 93.1 Kg/cm^{2}g (1325 psig), as measured by the reactor steam dome pressure indicator, is equivalent to 96.7 Kg/cm^{2}g (1375 psig) at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, 1974 Edition [later Edition] (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 87.9 Kg/cm^{2}g (1250 psig) for suction piping and 116 Kg/cm^{2}g (1650 psig) for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 87.9 Kg/cm^{2}g (1250 psig) for suction piping and 105.5 Kg/cm^{2}g (1500 psig) for discharge piping. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 97.7 Kg/cm^{2}g (1375 psig).

96.7

APPLICABILITY

SL 2.1.2 applies in all MODES; however, in MODE 5, because the reactor vessel head closure bolts are not fully tightened, it is unlikely the RCS would be pressurized.

(continued)

BASES (continued)

LCOs

LCO 3.0.3 (continued)

11 hours, because the total time for reaching MODE 3 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, and 3, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 4 and 5 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.6, "Fuel Pool Water Level." LCO 3.7.6 has an Applicability of "During movement of irradiated fuel assemblies in the associated fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.6 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.6 of "Suspend movement of irradiated fuel assemblies in the associated fuel storage pool(s)" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a different MODE or other specified condition when the following exist:

a. The requirements of an LCO, in the MODE or other specified condition to be entered, are not met; and

(continued)

BASES (continued)

LCOs

LCO 3.0.6 (continued)

Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.8, "Safety Function Determination Program" (SFDP), ensures loss of safety function is detected and appropriate actions are taken. Upon failure to meet two or more LCOs concurrently, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be:

a. $\geq 0.38\%\ \Delta k/k$, with the highest worth control rod or rod pair analytically determined; or

b. $\geq 0.28\%\ \Delta k/k$, with the highest worth control rod or rod pair determined by test.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1 Initiate action to ~~F~~fully insert all insertable control rods.	~~1 hour~~ Immediately
D. SDM not within limits in MODE 4.	D.1 Initiate action to ~~F~~fully insert all insertable control rods. AND	~~1 hour~~ Immediately (continued)

Table 3.1.4-1
Control Rod Scram Times

-----------------------------------NOTES-----------------------------------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Control rods with scram times > [ ] seconds to 60% rod insertion position are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

ROD POSITION PERCENT INSERTION (%)	SCRAM TIMES[a] (seconds)		
	REACTOR STEAM DOME PRESSURE[b] 0 Kg/cm^2g (0 psig)	REACTOR STEAM DOME PRESSURE[b] 66.8 Kg/cm^2g (950 psig)	REACTOR STEAM DOME PRESSURE[b] 73.8 Kg/cm^2g (1050 psig)
10	(c)	[ ]	[ ]
40	(c)	[ ]	[ ]
60	[ ]	[ ]	[ ]

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero.

intermediate

(b) For immediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.

(c) For reactor steam dome pressure $\leq$ 66.8 Kg/cm^2g (950 psig), only 60% rod insertion position scram time limit applies.

BASES

ACTIONS (continued)

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 within 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

immediately initiate action to

Action must continue until all insertable control rods are fully inserted.

With SDM not within limits in MODE 3, the operator must fully insert all insertable control rods~~, within 1 hour~~. This action results in the least reactive condition for the core. The allowed Completion Time of 1 hour is acceptable, considering the reactor can still be shut down, assuming no failures of additional control rods to insert.

e

D.1, D.2, D.3, and D.4

immediately initiate action to fully

With SDM not within limits in MODE 4, the operator must insert all insertable control rods. ~~in 1 hour~~ This action results in the least reactive condition for the core. The 1 hour Completion Time provides sufficient time to take corrective action and is acceptable, considering the reactor can still be shut down assuming no failures of additional control rods to insert. Actions must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment (LCO 3.6.4.1, "Secondary Containment") is OPERABLE; at least one Standby Gas Treatment (SGT) (LCO 3.6.4.3, "Standby Gas Treatment (SGT) System") subsystem is OPERABLE; and at least one secondary containment isolation valve (LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)") and associated instrumentation (LCO 3.3.6.1, "Isolation Instrumentation") are OPERABLE in each associated penetration flow path not isolated. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the SRs needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be

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(continued)

BASES

LCO
(continued)

determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are only allowed to be withdrawn under Special Operations LCO 3.10.3, "Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4, "Control Rod Withdrawal—Cold Shutdown," which provide adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Rod OPERABILITY—Refueling."

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

<u>A.1, A.2, and A.3</u>

3.3.2.1

A control rod is considered stuck if it will not insert by either FMCRD drive motor torque or scram pressure. The failure of a control rod to insert during SR 3.1.3.2 or SR 3.1.3.3 alone, however, does not necessarily mean that the control rod is stuck, since failure of the motor drive would also result in a failure of these tests. Verification of a stuck rod can be made by attempting to withdraw the rod. If the motor is working and the rod is actually stuck, the traveling nut will back down from the bottom of the drive and a rod separation alarm and rod block will result (see LCO 3.3.1). Conversely, if the motor drive is known to be failed, the rod is not necessarily inoperable since it is

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests. Required Action A.2 is modified by a Note that states the requirement is not applicable when below the actual low power setpoint (LPSP) of the RC&IS, since the notch insertions may not be compatible with the requirements of 3.3.1.2
rod pattern control (LCO 3.1.6) and the RC&IS (LCO 3.3.1).

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion an additional control rod would have to be assumed to have failed to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

With a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 6). Required action A.2 performs a step test on each remaining withdrawn control rod to ensure that no additional control rods are stuck. Therefore, 72 hours is allowed to perform the analysis to test in Required Action A.3.

B.1 and B.2

With two or more withdrawn control rods stuck, the stuck control rods should be isolated from scram pressure within 1 hour and the plant brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 66.8 Kg/cm^2g (950 psig). Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 will be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and a high pressure test may be required. This testing ensures that the control rod scram performance is acceptable for operating reactor pressure conditions prior to withdrawing the control rod for continued operation. Alternatively, a test during hydrostatic pressure testing could also satisfy both criteria.

The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability of testing the control rod at the different conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. ABWR SSAR, Section 4.6.2.
3. ABWR SSAR, Section 5.2.2.
4. ABWAR, Section 15.4.1.

BASES

ACTIONS

A.1 (continued)

result in requiring the affected control rod or rod pair to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3. The allowed Completion Time of 8 hours is considered reasonable, based on the large number of control rods available to provide the scram function. Additionally, an automatic reactor scram function is provided on sensed low pressure in the CRD charging water header (see LCO 3.3.1.1, "RPS Instrumentation"). This anticipatory reactor trip protects against the possibility of significant pressure degradation (and thus reduced scram force) concurrently in multiple control rod scram accumulators due to a transient in the CRD hydraulic system.

B.1

With two or more control rod scram accumulators inoperable, the scram function could become severely degraded because the accumulators are the primary source of scram force for the control rods at all reactor pressures. In this event, the associated control rods are declared inoperable and LCO 3.1.3 entered. This would result in requiring the affected control rods to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 1 hours is considered reasonable, based on the capability to drive in the control rods by the FMCRD motors and the low probability of a DBA or transient occurring while the affected accumulators are inoperable.

C.1

The reactor mode switch must be immediately placed in the shutdown position if any Required Action and associated Completion Time cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the Required Action is not applicable if all control rods associated with the

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} is within ± 1% $\Delta k/k$ during operations in MODE 1.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement. AND 1000 MWD/T thereafter

BASES

ACTIONS

A.1 (continued)

conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

This compar. requires the c.. to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain ~~meaning~~ meaningful results. Therefore, the comparision is only done when in MODE 2.

Verifying the reactivity difference between the monitored and predicted core k_{eff} is within the limits of the LCO provides further assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The Core Monitoring System calculates the core k_{eff} for the reactor conditions obtained from plant instrumentation. A comparison of the monitored core k_{eff} to the predicted core k_{eff} at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control

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